ACCESSION #: 9606040024

LICENSEE EVENT REPORT (LER)

FACILITY NAME: Edwin I. Hatch Nuclear Plant - Unit 1 PAGE: 1 OF 6

DOCKET NUMBER: 05000321

TITLE: Reactor Pressure Increase Results in Automatic Reactor

Shutdown

EVENT DATE: 04/30/96 LER #: 96-008-00 REPORT DATE: 05/29/96

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 011

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:

50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:

NAME: Steven B. Tipps, Nuclear Safety TELEPHONE: (912) 367-7851

and Compliance Manager, Hatch

COMPONENT FAILURE DESCRIPTION:

CAUSE: SYSTEM: COMPONENT: MANUFACTURER:

REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On 4/30/96 at 0841 EST, Unit 1 was in the Run mode at a power level of 280 CMWT (11% rated thermal power). At that time, the main turbine bypass valves closed unexpectedly causing reactor pressure to increase. The reactor shut down automatically on high pressure. Reactor water level decreased to 20.5 inches above instrument zero (179 inches above the top of the active fuel) before being restored by the reactor feedwater system. Reactor water level remained above the automatic reactor shutdown and Primary Containment Isolation System setpoint; therefore, no additional automatic

actions occurred. Pressure reached its maximum value of 1081 psig two seconds after the automatic shutdown and was reduced by steam loads. No safety/relief valves lifted nor were any required to lift.

This event was caused by an inadequate procedure. Procedure 34SO-N32-001-1S, "EHC Hydraulic System," contained instructions intended to isolate a portion of the Electrohydraulic Control (EHC) system while maintaining bypass valve capability. However, as was discovered after the event, the configuration of the EHC system piping does not allow a portion of the system to be isolated while maintaining bypass valve capability. When this subsection of the procedure was performed to repair EHC system leaks on the main turbine stop and control valves, EHC system fluid was partially isolated to the bypass valves and they failed closed per design. The procedure has been revised to eliminate the use of the incorrect instructions.

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PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor

Energy Industry Identification System codes appear in the text as (EIIS

Code XX).

DESCRIPTION OF EVENT

On 4/30/96 at 0841 EST, Unit 1 was in the Run mode at a power level of

280 CMWT (11% rated thermal power). At that time, the main turbine

bypass valves (EIIS Code SO) closed unexpectedly causing reactor pressure

to increase. Pressure exceeded the automatic high reactor pressure

shutdown nominal setpoint of 1080 psig and the reactor automatically shut

down per design.

Reactor water level decreased due to void collapse from the rapid

reduction in power. However, because the reactor was at low power at the

time of the automatic shutdown, the transient did not result in water

level decreasing to the automatic reactor shutdown and Group 2 Primary

Containment Isolation System (EIIS Code JM) setpoint of three inches above instrument zero. Therefore, no additional automatic actions occurred. Reactor water level decreased to its minimum value of 20.5 inches above instrument zero (179 inches above the top of the active fuel) before being restored automatically by the operating reactor feedwater pump (EIIS Code SJ). No Emergency Core Cooling Systems actuated nor were any required to actuate to restore or maintain water level.

Reactor pressure reached its maximum value of about 1081 psig two seconds after the automatic reactor shutdown. Pressure was reduced by steam loads, such as the reactor feedwater pump turbine (EIIS Code SJ), the steam sealing system (EIIS Code TC), and the steam jet air ejector (EIIS Code SH). No safety/relief valves lifted nor were any required to lift to reduce or control reactor pressure.

CAUSE OF EVENT

This event was caused by an inadequate procedure. Procedure 34SO-N32-001-1S, "EHC Hydraulic System," contained instructions intended to isolate a portion of the Electrohydraulic Control (EHC) system while maintaining bypass valve capability. However, as was discovered after the event, the configuration of the EHC system piping does not allow a portion of the system to be isolated while maintaining bypass valve capability. When this subsection of the procedure was performed on 4/30/96 to isolate and repair EHC system leaks on the main turbine stop and control valves (EIIS Code TA), EHC system fluid was partially isolated to the bypass valves and they failed closed per design.

The EHC system supplies high pressure hydraulic fluid to the actuators for the main turbine bypass, stop, control valves, and combined intercept valves for the purpose of moving the valves. The fluid supplied to the actuators is termed the Fluid Actuator Supply (FAS). The EHC system also supplies high pressure hydraulic fluid to the servovalves for the bypass, stop, control, and combined intercept valves. This fluid, termed the Fluid Jet Supply (FJS), positions a spool piece within the servovalve which in turn directs FAS fluid to the valves to open or close them. When the FJS is lost, the affected valve fails closed. In the plant's original design, the FAS and the FJS were routed to the valves via separate lines running from a common header.

In the late 1970's, the EHC system design was changed per General Electric Turbine Information Letter 841-3A. Part of the recommended change was the elimination of the separate FJS lines. The FJS was to be supplied from the FAS lines by use of a manifold added at the servovalve. The manifold would re-route a portion of the FAS to the FJS port of the servovalve and eliminate the need for a separate FJS line to the servovalve. However, this proposed change to the configuration of the fluid supply lines to the three bypass valves was never implemented in the plant. A review of plant maintenance records subsequent to this event was unable to locate documentation supporting the retention of the existing configuration.

In 1984, another change was made to the EHC system design as recommended by General Electric. Design Change Request 83-174 re-routed some of the bypass valve FAS lines and added two isolation valves. The purpose of this change was to allow the bypass valves to continue to be supplied with FAS fluid, and therefore to continue to operate, while the hydraulic fluid supplies to the control, stop, and combined intercept valves were isolated. However, the design change was implemented based upon the incorrect assumption that the FJS to the bypass valves was supplied by their FAS line. Consequently, only the FAS line to the bypass valves was re-routed to a header apart from the fluid supply to the control, stop, and combined intercept valves. The intent of the design change was not met in that the fluid supplies to the control, stop, and combined intercept valves could not be isolated without also isolating the FJS to the bypass valves since it remained separate from the FAS to the bypass valves. Therefore, after implementation of the DCR, isolation of the FJS continued to prevent the bypass valves from operating.

In 1987, procedure 34SO-N32-001-1S was revised to incorporate the changes made by Design Change Request 83-174. The procedure change also incorporated the incorrect assumption that the FJS fluid to the bypass valves was supplied by their FAS line. The procedure included instructions to isolate the portion of the EHC system going to the control, stop, and combined intercept valves while leaving in service the FAS line to the bypass valves. Because the FJS fluid to the bypass valves is actually supplied by the separate FJS lines, the FJS is also isolated when the portion of the EHC system going to the control and stop valves is isolated. The bypass valves subsequently fail closed.

In November 1988, vendor drawings showing the incorrect configuration of the EHC system were converted to GPC drawings. After the conversion process, the GPC drawings accurately reflected EHC system design in that the FJS to the bypass valves was shown as a separate line and not as being supplied by the FAS line to the bypass valves or from the independent FAS header installed by Design Change Request 83-174. However, procedure 34SO-N32-001-1S was not subsequently revised to accurately reflect the correct EHC system piping configuration as shown on the GPC drawings. Therefore, the procedure contained incorrect instructions for the isolation of the fluid supply to the bypass, control, stop, and combined intercept valves. Specifically, the procedure contained instructions which were based upon an incorrect system configuration which would allow the fluid supply to the control, stop, and combined intercept valves to be isolated without affecting the ability of the bypass valves to operate.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This report is required by 10 CFR 50.73 (a)(2)(iv) because of the unplanned actuation of Engineered Safety Feature systems. The Reactor Protection System (EIIS Code JC), an Engineered Safety Feature system, actuated on high reactor pressure per design when pressure increased to approximately 1081 psig following the closure of the bypass valves.

An increase in pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This causes the neutron flux and thermal power transferred to the reactor coolant to increase which could challenge the fuel thermal limits and the reactor coolant pressure boundary. Therefore, the reactor is shut down automatically on high reactor steam dome pressure to limit the neutron flux and thermal power increase. The automatic reactor shutdown on high pressure, along with the safety/relief valves, limits the peak reactor pressure to less than the American Society of Mechanical Engineers Section III Code limits.

In this event, the bypass valves closed per design when EHC system fluid to the valves was partially isolated during the performance of a subsection of procedure 34SO-N32-001-1S. When the valves closed, reactor pressure increased; when pressure reached approximately 1080 psig, the reactor automatically shut down per design. Pressure reached its maximum value of about 1081 psig two seconds after the automatic shutdown and was reduced by steam loads. No safety/relief valves opened nor were any required to open to limit or reduce pressure.

Reactor water level decreased due to void collapse from the rapid decrease in reactor power. The operating reactor feedwater pump automatically restored water level; the minimum water level reached was 179 inches above the top of the active fuel and was only about 16.5 inches below normal. No Emergency Core Cooling Systems actuated nor were any required to actuate to recover or maintain water level during or following this event. All automatic functions operated per design in response to the pressure increase and the automatic reactor shutdown.

Based upon the preceding discussion, it is concluded that this event had no adverse impact on nuclear safety. This analysis is applicable to all power levels.

CORRECTIVE ACTIONS

Procedures 34SO-N32-001-1S and 34SO-N32-001-2S, "EHC Hydraulic System," were revised to eliminate the instructions permitting the isolation of the portion of the EHC systems going to the main turbine control and stop valves. Procedure 34SO-N32-001-2S was revised as a precaution until such time as the configuration of the FJS and the FAS to the Unit 2 bypass valves can be verified.

ADDITIONAL INFORMATION

No systems other than those already mentioned in this report were affected by this event.

No failed components caused or resulted from this event.

Previous similar events in which an unplanned automatic reactor shutdown on high pressure occurred were reported in Licensee Event Reports 50-321/1996-001, dated 1/26/96, and 50-321/1996-004, dated 4/15/96. In the first event, high pressure resulted when the control valves drifted closed while the unit was operating at greater than 30% power. The valves drifted closed due to material blocking their servovalve strainers preventing hydraulic fluid from reaching the servovalve spools. Upon loss of fluid, the control valves closed per their fail-safe design. As the control valves drifted closed, the bypass valves opened to control pressure. However, the bypass valves could not pass all the steam generated at the high power level. Therefore, when the control valves closed, the excessive steam generated caused pressure to increase and the reactor automatically shutdown.

In the second event, the control and stop valves closed rapidly when the turbine was shut down manually to begin a refueling outage. Their rapid closure resulted in a pressure increase of about 50 psig. The bypass valves opened to limit the pressure increase. However, there was insufficient margin between the pressure at the time the main turbine was shut down manually and the automatic high reactor pressure shutdown setpoint. When pressure increased to the high pressure setpoint, the reactor automatically shut down per design.

In this event, the bypass valves closed when EHC system fluid to the valves was partially isolated causing the valves to fail closed per design. Since the causes of the three events were different, the

corrective actions for the previous events could not have anticipated or

prevented this event.

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Georgia Power Company

40 Inverness Center Parkway

Post Office Box 1295

Birmingham, Alabama 35201

Telephone 205 877-7279

J. T. Beckham, Jr. Georgia Power

Vice President - Nuclear the southern electric system

Hatch Project

May 29, 1996

Docket No. 50-321 HL-5177

U.S. Nuclear Regulatory Commission

ATTN: Document Control Desk

Washington, D.C. 20555

Edwin I. Hatch Nuclear Plant - Unit 1

Licensee Event Report

Inadequate Procedure Results in Reactor

Pressure Increase and Automatic Reactor Shutdown

Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(iv), Georgia Power Company is submitting the enclosed Licensee Event Report (LER) concerning the inadequate procedure which resulted in a reactor pressure increase and automatic reactor shutdown.

Sincerely,

J. T. Beckham, Jr.

SMS/eb

Enclosure: LER 50-321/1996-008

cc: Georgia Power Company

Mr. H. L. Sumner, General Manager - Nuclear Plant

NORMS

U.S. Nuclear Regulatory Commission, Washington, D.C.

Mr. K. Jabbour, Licensing Project Manager - Hatch

U.S. Nuclear Regulatory Commission, Region II

Mr. S. D. Ebneter, Regional Administrator

Mr. B. L. Holbrook, Senior Resident Inspector - Hatch

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